Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

John T. Herron
Vice President, Browns Ferry Nuclear Plant

June 22, 2000

U.S. Nuclear Regulatory Commission 10 CFR 50.73
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

In the Matter of) Docket No. 50-296
Tennessee Valley Authority)

BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 3 - DOCKET NO. 50-296 - FACILITY OPERATING LICENSE DPR-68 - LICENSEE EVENT REPORT (LER) 50-296/2000-005-00

The enclosed report provides details of the May 24, 2000, automatic reactor scram on Unit 3 from 100 percent power. The scram was the result of an invalid low reactor water level signal generated while returning a feedwater level transmitter to service following scheduled calibration. All plant safety systems operated as designed in response to this event.

This report is submitted in accordance with 10 CFR 50.73 (a)(2)(iv) as an event that resulted in the actuation of an engineered safety feature, including the reactor protection system.

Sincerely,

John T. Herron

cc: See page 2

IE22

Enclosure
cc (Enclosure):
Mr. William O. Long, Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

Mr. Paul E. Frederickson, Branch Chief
U.S. Nuclear Regulatory Commission
Region II
61 Forsyth Street, S. W.
Suite 23T85
Atlanta, Georgia 30303

NRC Resident Inspector
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, Alabama 35611

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Browns Ferry Nuclear Plant Unit 3	05000296	1 of 7

TITLE (4)
Scram During Level Transmitter Calibration

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER				FACILITY NAME	DOCKET NUMBER
									NA	
05	24	00	2000	005	000	06	22	00	NA	DOCKET NUMBER

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)									
POWER LEVEL (10)	100		20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(viii)
			20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Anthony T. Rogers, Senior Licensing Project Manager	(256) 729-2977

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On May 24, 2000, with Unit 3 operating at 100 percent power, an invalid low reactor water level scram signal was generated while returning a feedwater level transmitter to service following scheduled calibration. The reactor scram caused reactor water level to decrease below the low level (level 3) and low-low level (level 2) setpoints. All emergency systems operated as expected in response to the scram, including insertion of all control rods. This event did not result in loss of the normal heat removal path as described in NEI 99-02, Revision 0, since the condenser remained available throughout the event and was used for decay heat removal.

The root cause of this event was determined to be lack of specific proceduralized valving sequences for this level transmitter based on the current design and procedural guidance. The scram was the result of a pressure perturbation in the common variable sensing line shared with both channels of reactor protection system level transmitters. Corrective actions include revising the applicable procedure.

TVA is reporting this event in accordance with 10 CFR 50.73 (a)(2)(iv) as an event that resulted in an actuation of an engineered safety feature, including the reactor protection system.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION	
Browns Ferry Nuclear Plant - Unit 3	05000296	2000	-- 005	-- 000	2 of 7

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

I. PLANT CONDITIONS

At the time of this event, Unit 2 and Unit 3 were operating in Mode 1 at 100 percent power. Unit 1 was shutdown and defueled.

II. DESCRIPTION OF EVENT

A. Event:

On May 24, 2000, at 1354 hours Central Daylight Time (CDT) with Unit 3 operating at 100 percent power, an invalid low reactor water level scram signal was generated while returning a feedwater level transmitter to service following scheduled calibration. All emergency systems operated as expected in response to the scram, including insertion of all control rods.

The scram resulted in the expected automatic actuation or isolation of the following PCIS [JE] systems and components at the low level (level 3) setpoint:

- PCIS group 2, Shutdown cooling mode of Residual Heat Removal (RHR) [BO] system; drywell floor drain isolation valves; drywell equipment drain isolation valves [WP].
- PCIS group 3, Reactor Water Cleanup (RWCU) [CE] system.
- PCIS group 6, primary containment purge and ventilation [JM], Unit 3 reactor zone ventilation [VB]; refuel zone ventilation [VA]; Standby Gas Treatment system [BH]; Control Room Emergency Ventilation system [VI].
- PCIS group 8, Traversing Incore Probe (TIP) [IG].

Following the reactor scram, reactor water level decreased below the low-low level (level 2) setpoint, due to the void collapse and response of the Feedwater Control System (FCS). High Pressure Coolant Injection (HPCI) [BG] and Reactor Core Isolation Cooling (RCIC) [BN] systems both initiated at the low-low level setpoint and injected. The low-low level signal also tripped both recirculation pumps [AD] and actuated the Alternate Rod Injection (ARI) systems.

TVA is reporting this event in accordance with 10 CFR 50.73 (a)(2)(iv) as an event that resulted in an actuation of an engineered safety feature, including the reactor protection system.

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

None.

C. **Dates and Approximate Times of Major Occurrences:**

May 24, 2000, at 1354 hours CDT	Unit 3 received an invalid low reactor water level scram signal while returning a feedwater level transmitter to service following calibration. Both recirculation pumps tripped. HPCI and RCIC initiated and recovered water level.
May 24, 2000, at 1403 hours CDT	Scram and PCIS reset. Normal ventilation restored to service.
May 24, 2000, at 1405 hours CDT	Returned HPCI and RCIC to standby readiness.
May 24, 2000, at 1408 hours CDT	Placed 3A Recirculation Pump in service.
May 24, 2000, at 1410 hours CDT	Standby Gas Treatment and Control Room Emergency Ventilation systems secured.

D. **Other Systems or Secondary Functions Affected:**

None.

E. **Method of Discovery:**

The operating crew in the main control room immediately recognized the automatic scram had occurred.

F. **Operator Actions:**

Operator actions in response to the event were proper and in accordance with applicable plant procedures.

G. **Safety System Response:**

All required safety systems responded as designed.

III. CAUSE OF THE EVENT

A. **Immediate Cause:**

The immediate cause was a pressure perturbation in the sensing line to reactor water level instruments that sensed a low reactor water level in both channels of the reactor protection system (RPS).

B. **Root Cause:**

The root cause of this event was determined to be lack of a specific proceduralized valving sequence for this level transmitter to reduce the risk of inducing a perturbation into the reactor instrument sensing lines.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION	4 of 7
Browns Ferry Nuclear Plant - Unit 3	05000296	2000	-- 005	-- 000	

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

C. **Contributing Factors:**

None.

IV. ANALYSIS OF THE EVENT

The reactor scram occurred while Instrument Mechanics were returning a feedwater level transmitter, 3-LT-3-206 (Foxboro), to service following scheduled calibration. This transmitter shares a common sensing line (variable leg) with other reactor narrow range level transmitters (Rosemount) that input to the RPS (see Attachment 1). Due to the type of sensing element used on the Foxboro transmitters, their process connections are reverse that of the Rosemount transmitters. By design, the RPS transmitters use separate reference sensing lines but share a common variable sensing line. This feature combined with a specific valving sequence included in RPS procedures, but not in the feedwater transmitter calibration procedure, prevents a full scram in the event of a pressure perturbation while returning a transmitter to service. When 3-LT-3-206 was being returned to service, its low side manifold isolation valve (variable leg) was first opened. This action would pressurize both the high and low sides of 3-LT-3-206 since the manifold equalizing valve was open. While the low side manifold isolation valve was being opened, the scram occurred. Data obtained from an adjacent transmitter, which shares the variable sense line with 3-LT-3-206, indicated reactor water level showed a substantial lowering. This is attributed to the variable sense line pressure lowering while reestablishing pressure to the tubing associated with 3-LT-3-206. Since both RPS transmitters share this same sense line, a momentary low reactor water level scram signal was generated.

Review of the sequence of events determined that no personnel performance issues contributed to this incident. No actions were found to be inappropriate for the work activity. The key personnel actions associated with the valve manipulation being made to return the 3-LT-3-206 to service when the reactor scram occurred were found to be in accordance with management expectations and training the craftsmen receive. Known cautions were discussed prior to performance of the activity. Self-checking practices were applied prior to performing the valve manipulations. Actions to be performed in support of instrument return to service were discussed and agreed upon prior to being performed, which is in accordance with the industry and plant standard practice of opening the variable leg isolation first unless specifically directed by procedure. Therefore, the valve manipulation were in accordance with training material content and documented appropriately.

During this event, reactor water level lowered below the HPCI and RCIC initiation setpoint, low-low level (level 2). The void collapse following the scram and response of the FCS contributed to the reactor water level lowering to the low-low setpoint. As a result, data pertaining to the FCS was reviewed to determine if any anomalies occurred. The preprogrammed response was initiated after the reactor scram and after the FCS detected the low water level confirmatory setpoint. As designed, feedwater pumps 3A and 3B automatically swap to Manual Control with a minimum speed demand and feedwater pump 3C remained in Automatic Control to control reactor water level. Therefore, the system responded as designed. This event did not result in loss of the normal heat removal path as described in NEI 99-02, Revision 0, since the condenser remained available throughout the event and was used for decay heat removal.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The evaluation of plant system and component responses to the event concluded that all emergency systems responded as designed. The normal heat removal path was not lost during this event since the condenser was used for decay heat removal and no main steam relief valves opened. Personnel performance in response to the scram was also evaluated and found to be satisfactory for an event of this type. In addition, there was no radioactive material released, and there were no actual or potential safety consequences as a result of this event.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions:

Calibration of the transmitter was completed satisfactorily and calibration of other Foxboro transmitters used for reactor water level on Unit 2 and Unit 3 were placed on hold until applicable corrective actions are completed. In addition, testing and maintenance activities which can cause a half-scram or engineered safety feature actuation have been identified for assessment prior to performance.

B. Corrective Actions to Prevent Recurrence:

The procedure used to calibrate the Foxboro reactor water level transmitters will be revised to contain specific valve sequencing.[1]

Procedures that calibrate instruments with common sensing lines that could cause a reactor scram or turbine trip will be evaluated, revised or rescheduled as necessary to preclude a similar occurrence. [1]

A design modification will be issued for anti-surge tip isolation valves to be installed on Unit 3 where the Foxboro reactor water level transmitters are used. An evaluation will be conducted for installation of anti-surge tip isolation valves for other transmitters that have common sensing lines which can generate a reactor scram either directly or indirectly. [1]

The response of the FCS will be evaluated to determine if initiation of HPCI and RCIC can be avoided for this type of transient. [1]

---

[1]TVA does not consider this corrective action a regulatory commitment. The completion of this item will be tracked in TVA's Corrective Action Program.

A. Failed Components:

None.

B. Previous Similar Events:

No corrective actions for any previous similar event could have prevented this event.

C. Additional Information:

This event did not result in loss of the normal heat removal path as described in NEI 99-02, Revision 0, since the condenser remained available throughout the event and was used for decay heat removal.

D. Safety System Functional Failure:

This event did not result in a safety system functional failure in accordance with NEI 99-02, Revision 0.

VIII. COMMITMENTS

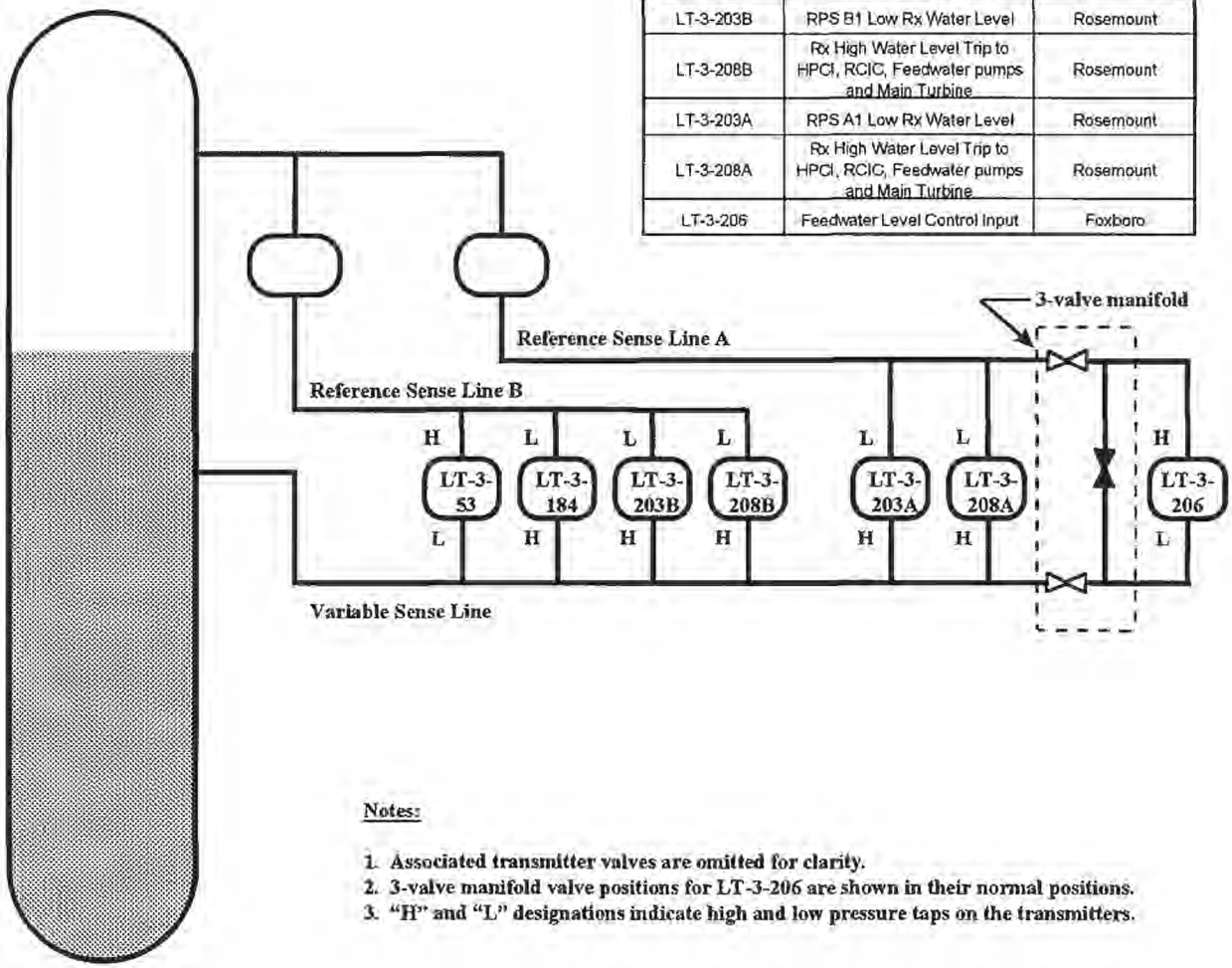
None.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION	
Browns Ferry Nuclear Plant - Unit 3	05000296	2000	- 005	- 000	7 of 7

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

ATTACHMENT 1
Reactor Vessel Level Transmitter Arrangement
(Half of all transmitters shown for clarity)

Transmitter ID	Function	Transmitter Type
LT-3-53	Feedwater Level Control Input	Foxboro
LT-3-184	ADS Confirmatory Logic	Rosemount
LT-3-203B	RPS B1 Low Rx Water Level	Rosemount
LT-3-208B	Rx High Water Level Trip to HPCI, RCIC, Feedwater pumps and Main Turbine	Rosemount
LT-3-203A	RPS A1 Low Rx Water Level	Rosemount
LT-3-208A	Rx High Water Level Trip to HPCI, RCIC, Feedwater pumps and Main Turbine	Rosemount
LT-3-206	Feedwater Level Control Input	Foxboro



Notes:

1. **Associated transmitter valves are omitted for clarity.**
2. **3-valve manifold valve positions for LT-3-206 are shown in their normal positions.**
3. **"H" and "L" designations indicate high and low pressure taps on the transmitters.**